ACCESSION #: 9604050071

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Oconee Nuclear Station, Unit One PAGE: 1 OF 10

DOCKET NUMBER: 05000269

TITLE: Loss of Feedwater Results In Reactor Trip Due To

Equipment Failure

EVENT DATE: 02/28/96 LER #: 96-04-00 REPORT DATE: 03/28/96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: L. V. Wilkie, Safety Review TELEPHONE: (864) 885-3518

Maanger

COMPONENT FAILURE DESCRIPTION:

CAUSE: F SYSTEM: JA COMPONENT: IMOD MANUFACTURER: B045

F SD PDCO F130

REPORTABLE NPRDS: Yes

Yes

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On February 28, 1996, at approximately 2102 hours, while operating at 100% full power, Unit 1 experienced a Feedwater transient. Operators observed Reactor Coolant System pressure and average temperature increasing. The Integrated Control System (ICS) was

taken to manual in an attempt to stabilize the unit during the transient. The Reactor tripped from approximately 95% full power at approximately 2103 hours due to the loss of both Main Feedwater pumps on low suction pressure. The Emergency Feedwater pumps started and maintained Steam Generator levels. The unit was stabilized at hot shutdown conditions. Investigations indicated the cause of the initiating Feedwater transient was a failed multiplier card in the ICS and the cause of the Reactor trip was a failed positioner relay on a valve in the Condensate system. The root cause of the event is an equipment failure as the result of a degraded subcomponent. Corrective actions included replacing the ICS multiplier card and the positioner relay on the Condensate system valve.

BACKGROUND

The High Pressure Injection (HPI) [EIIS:BG] System performs a dual function. First, it supplies normal Reactor Coolant System (RCS) [EIIS:AB] pump seal injection, and make-up for inventory and chemistry control. Second, it is the Engineered Safeguards (ES) [EIIS:JE] high pressure safety injection system and part of the Emergency Core Cooling System. Normal post-trip response includes RCS density and volume changes due to reduction of the RCS average temperature from 579 F to 555 F. The volume change can result in uncovering the Pressurizer (PZR) heaters unless make-up flow is increased to compensate. Typically the flow rate exceeds the capacity of the normal make-up control valve (HP-120) so that a parallel valve (HP-26) is manually opened. The increased make-up flow may divert flow from the reactor coolant pump seal injection flow path, in which case the low seal flow signal will auto start a second HPI pump. After a short duration, typically one to five minutes, the PZR level is restored, HP-26 is closed to reduce make-up flow to normal, and the second HPI pump is secured. These actions are included in the Emergency Operating Procedure and are considered to be a preplanned normal post-trip response. This sequence is not considered a manual or automatic actuation of the ES function.

The Integrated Control System (ICS) [EIIS:JA] provides fully automatic control of reactor power, steam generation rate, and generated load by processing selected signals of measured plant parameters.

The Integrated Master portion of the ICS develops a Steam Generator (SG) Demand signal based on the Unit Load Demand for generated Megawatts. The SG Demand signal represents the required Feedwater (FDW) [EIIS:SJ] flow to the SGs necessary to produce the amount of steam needed by the Main Turbine. These signals are input to a multiplier module and function generator. The output of the multiplier is fed to a summer module. This summer module also receives a RCS average temperature (Tave)

error input and a Neutron error input. The output of the summer module is the FDW Master Demand signal.

The Condensate System [EIIS:SD] originates at the Condenser Hotwell. The Hotwell Pumps supply Condensate to the Polishing Demineralizers, Condensate Coolers (CC), Generator Hydrogen and Water Coolers (GHWC), and Air Ejectors before entering the Condensate Booster Pumps (CBP). The CBPs increase system pressure to that required for the Main Feedwater Pump net positive suction head.

There is a bypass line around the CCs and GHWCs to ensure that the remaining components in the system have adequate Condensate flow. An air operated valve (1C-61) controls the flow based on the differential pressure (D/P) across the GHWCs. If the D/P increases to a preset value, the controller "trips" the valve to the full open position.

EVENT DESCRIPTION

On February 28, 1996, at 2100 hours, Unit 1 was operating at 100% full power. There were no tests or major work in progress. At approximately 2102 hours, computer alarms were received and acknowledged in the control room for "FDWP A/B Seal Diff Press Lo" and "Stm Gen B BTU Limit". These alarms cleared and recurred several times. A Feedwater (FDW) transient began causing Reactor Coolant System (RCS) pressure and average temperature (Tave) to increase. Group 7 control rods began inserting, the third Condensate Booster Pump (CBP) started, and operators placed the appropriate portions of the Integrated Control System (ICS) to manual in an attempt to stabilize the Unit.

At approximately 2103 hours, all three CBPs tripped causing both Main Feedwater Pumps (MFDWP) to trip on low suction pressure. ' The Reactor tripped at 2103:09 hours on the anticipatory Reactor Protective System [EIIS:JC] trip signal from the loss of both MFDWPs.

Several immediate automatic actions occurred. The Control Rod Drive [EIIS:AA] breakers opened, and all full length control rods inserted into the core, shutting down the Reactor. The Main Steam Relief Valves and

Turbine Bypass Valves opened. Both Motor Driven Emergency Feedwater Pumps (MDEFDWP) and the Turbine Driven Emergency Feedwater Pump (TDEFWP) started.

The operators also took manual action per the Emergency Operating Procedure (EOP). They confirmed that the Reactor and Main Turbine had tripped, verified that the Emergency Feedwater Pumps had started, and monitored for proper operation of other automatic equipment. They opened valve 1HP-26, as directed by the EOP, to avoid the Pressurizer (PZR) level dropping below the PZR heaters. This pre-planned, normal, action automatically started a second High Pressure Injection (HPI) Pump at approximately 2104 hours. At approximately 2109 hours, the operators stopped the second HPI pump and closed valve 1HP-26 when PZR level reached the normal post-trip setpoint.

Valve 1MS-77 (Main Steam (MS) to the 2nd stage Moisture Separator Reheater) failed to close when the unit tripped and caused MS pressure, in the B header, to decrease slightly. At approximately 2111 hours an upstream valve was closed by the operators and MS pressure returned to the normal post-trip condition.

The operators secured the TDEFWP at 2114 hours, after confirming that both MDEFDWPs were operating and supplying the Steam Generators (SG). Specific post-trip parameters remained within acceptable limits. RCS pressure increased to 2335 psig during the transient and decreased to 1816 psig after the Reactor trip. RCS pressure then slowly returned to

2128 psig. Pressurizer inventory remained on scale between a high of 266 inches immediately prior to the trip to a low of 81 inches post trip before increasing and stabilizing

at 144 inches. RCS average temperature (Tave) increased to approximately 585 F before the trip due to the transient, then decreased to approximately 547 F before increasing and stabilizing at 555 F.

Immediately following the trip, the 1A and 1B SG pressures reached a post trip high of approximately 1123 and 1122 psig, respectively. The pressures then decreased to 982 and 929 psig, respectively. The 1B SG pressure decreased to 929 psig due to valve 1MS-77 remaining open. After the 1A and 1B SG pressures returned to approximately 1010 psig, the pressure was reduced to 970 psig to close one of the MS Relief Valves (1MS-5) in the 1A header.

An investigation into the cause of the FDW transient and the unit trip was initiated. The SGs were supplied by the MDEFDWPs until the cause of unit trip was determined.

On February 29, 1996, at 0503 hours, both MDEFDWPs were secured after troubleshooting had identified the cause of the loss of MFDW and MFDWPs were returned to service.

For the FDW transient, Maintenance personnel concluded that the problem was limited to the circuitry that develops the FDW Master Demand signal in the ICS. This conclusion was based on the SG Demand signal and FDW temperature signal being steady; however, the FDW Master Demand signal was erratic just prior to the unit trip. The troubleshooting identified three modules in the circuitry which could exhibit this type of condition (summer, function generator, and multiplier). These three modules in the FDW Master Demand circuit were checked for proper operation and no obvious abnormalities were found. However, as a conservative measure, the multiplier module, summer, and function generator were replaced. The multiplier module, being the most complex, was placed in a test rack and the output was observed over a two day period. voltage output of the module was observed to be erratic.

Initially, the FDW transient caused control valve 1C-61 to go in the closed direction. When the Control Room Operator placed the ICS portions in manual and increased demand, valve 1C-61 did not respond properly which resulted in CBPs and MFDW pumps tripping due to low suction pressure. The investigation indicated that the controller in the control room was functioning satisfactorily, but the valve trip signal from this controller did not trip open valve 1C-61 as required. The valve positioner calibrated satisfactorily, but it would not function properly when a trip signal from the controller was initiated. After further investigation by the technicians, it was determined that the pneumatic relay (relay that opens the valve) in the positioner was degraded and would not pass adequate flow. The calibration procedure did not specifically check the trip response; therefore, the degraded relay had not been identified during performance of the procedure. The relay was replaced. The procedure for the instrument calibration was revised and the valve was re-calibrated. This procedure verified valve 1C-61 would stroke fully open from a trip signal initiated at the controller.

Valve 1MS-77 did not close, when required. An investigation indicated that the thermal overloads were tripped and the valve was on the back seat. The investigation found that the motor was damaged and near failure. This resulted in the thermal overloads tripping. Also, the limit switch for the valve operator was not set properly. The procedure specifies the limit switch be set at 5% off the back seat; however, it was found at approximately 2%. The investigation concluded that the last time the operator was set, the limit was set at something less than 5% closed. The thermal overloads tripped when the valve was subsequently fully opened. The valve operator motor was replaced, the limit switch reset, and the valve tested satisfactorily. A Problem Investigation Process Report was initiated for investigation and corrective action associated with this problem.

The Plant Operations Review Committee was convened on February 29, 1996, at 1630 hours. The unit startup was approved pending the satisfactory completion of work on valve 1C-61 and the ICS Feedwater Temperature Compensation Circuit items (multiplier module, summer, and function generator).

On March 1, 1996, at 0737 hours, all the issues associated with the unit trip had been resolved and the Station Manager gave permission for startup. The unit was taken critical at 0939 hours and the generator placed on line at 1359 hours.

During the investigation of the Reactor trip, it was noted that problems with valve 1C-61 had been identified during the most recent refueling outage, ending in December 1995. Valve 1C-61 was removed to investigate a noise coming from the valve body. When the valve was inspected, technicians could not locate any problems. The valve was reinstalled and tested satisfactorily.

During the Unit 1 start-up from the refueling outage, valve 1C-61 was discovered to be stuck closed. operations personnel placed the valve controller in manual and the valve opened. When the controller was returned to automatic, the valve controlled properly.

CONCLUSIONS

The root cause of this event is Equipment Failure as the result of a degraded subcomponent.

The cause of the reactor trip is the failure of valve 1C-61 to respond as designed for a plant transient. The degraded pneumatic relay had not been found previously in post maintenance testing or preventive maintenance because the test procedure did not specifically challenge the relay trip function.

The failure in the Integrated Control System (ICS) circuit initiated the transient, but this should not have resulted in a reactor trip. The ICS is to be replaced with an upgraded system during the next scheduled refueling outage, for each unit, beginning with Unit 3 in 1996.

Unit 1 has not tripped since February 26, 1994. That previous trip (LER 269/94-02) was attributed to an inappropriate action. Root causes of other unit trips and other events over the past two years were also reviewed and indicated that the cause of the unit trip identified in this report is not a recurring problem.

The failure of the Integrated Control System (ICS) multiplier module and the positioner for the operator in valve 1C-61 are NPRDS reportable. The ICS multiplier module, model 6618210-1, is manufactured by Bailey Controls. The positioner relay is a subcomponent of a Fisher model number 476-U-15 valve operator.

There were no personnel injuries, radiation over-exposures, or releases of radioactive materials associated with this event.

CORRECTIVE ACTIONS

Immediate

1. The Integrated Control System (ICS) was placed in manual in response to the Feedwater (FDW) transient.

2. Operations personnel took appropriate actions in accordance with the Emergency Operating Procedure to bring the unit to

stable conditions following the Reactor trip.

Subsequent

1. Three modules in the ICS FDW Master Demand circuit were replaced.

2. Valve 1C-61 positioner relay was replaced and satisfactorily tested.

3. The valve operator motor on 1MS-77 was replaced, the limit switch reset to the proper back seat, and tested satisfactorily.

4. The procedure for calibration of valve 1C-61 was revised, based on the troubleshooting performed, during this event.

Planned

1. Determine other Feedwater/Condensate control valves which have similar design characteristics to valve 1C-61 and could result in unit trips. Review the testing/calibration procedures to assure that critical rate functions are verified.

2. Perform the revised calibration procedure on Unit 2 and 3 valves C-61.

Planned corrective actions 1 and 2 are considered Commitments to the NRC.

They are the only items included in this report intended to be NRC Commitments.

SAFETY ANALYSIS

A Feedwater transient was induced due to the failure of the multiplier module. The resulting Reactor trip was due to valve 1C-61 not responding to the demand, reducing the condensate flow to the Condensate Booster Pumps. This resulted in both Main Feedwater (MFDW) pumps tripping on low suction pressure.

TEXT PAGE 10 OF 10

Loss of MFDW is an anticipated transient and is described in Section 10.4 of the Final Safety Analysis Report. Loss of MFDW initiates a Reactor trip and starts the Emergency Feedwater (EFDW) System to provide decay heat removal. In this event, all the systems and equipment operated as designed to mitigate the consequences of the loss of MFDW. The MFDW pumps and Main Turbine (MT) tripped as expected. Instrumentation detected the loss of both MFDW pumps and the MT and initiated the Reactor trip and provided the start signal to the EFDW system. All three EFDW pumps started and the unit was stabilized at hot shutdown.

There were no releases of radioactive materials, radiation over-exposures, or personnel injuries associated with this event. The health and safety of the public was not affected by this event.

ATTACHMENT 1 TO 9604050071 PAGE 1 OF 2ATTACHMENT 1 TO 9604050071 PAGE 1 OF 2

Duke Power Company J. W. Hampton

Oconee Nuclear Generation Department Vice President

P.O. Box 1439 (803)885-3499 Office

Seneca, SC 29679 (803)885-3564 Fax

DUKE POWER

March 28, 1996

U.S. Nuclear Regulatory Commission

Document Control Desk

Washington, D.C. 20555

Subject: Oconee Nuclear Station

Docket Nos. 50-269, -270, -287

Licensee Event Report 269/96-04

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report, 269/96-03, concerning the loss of feedwater which resulted in a reactor trip.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

J. W. Hampton, Vice President

Oconee Nuclear Site

/fts

Attachment

ATTACHMENT 1 TO 9604050071 PAGE 2 OF 2

Document Control Desk

March 28, 1996

xc: Mr. L. A. Wiens, Project Manager

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

Washington, D.C. 20555

Mr. S. D. Ebneter, Regional Administrator

U.S. Nuclear Regulatory Commission

101 Marietta St., NW, Suite 2900

Atlanta, GA 30323

Mr. P. E. Harmon

NRC Resident Inspector

Oconee Nuclear Station

INPO Records Center

700 Galleria Parkway

Atlanta, GA 30339-5957

*** END OF DOCUMENT ***